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state and/or local radiological emergency plan that would in fact be relied upon in a radiological emergency.

(2) Generally, the plume exposure pathway EPZ for nuclear power plants shall consist of an area about 10 miles (16 km) in radius and the ingestion pathway EPZ shall consist of an area about 50 miles (80 km) in radius. The exact size and configuration of the EPZs surrounding a particular nuclear power reactor shall be determined in relation to local emergency response needs and capabilities as they are affected by such conditions as demography, topography, land characteristics, access routes, and jurisdictional boundaries. The size of the EPZs also may be determined on a case-by-case basis for gas-cooled nuclear reactors and for reactors with an authorized power level less than 250 MW thermal. The plans for the ingestion pathway shall focus on such actions as are appropriate to protect the food ingestion pathway.

(d) Notwithstanding the requirements of paragraphs (a) and (b) of this section, and except as specified by this paragraph, no NRC or FEMA review, findings, or determinations concerning the state of offsite emergency preparedness or the adequacy of and capability to implement State and local or utility offsite emergency plans are required prior to issuance of an operating license authorizing only fuel loading or low power testing and training (up to 5 percent of the rated power). Insofar as emergency planning and preparedness requirements are concerned, a license authorizing fuel loading and/or low power testing and training may be issued after a finding is made by the NRC that the state of onsite emergency preparedness provides reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency. The NRC will base this finding on its assessment of the applicant's onsite emergency plans against the pertinent standards in paragraph (b) of this section and appendix E. Review of applicant's emergency plans will include the following standards with offsite aspects:

(1) Arrangements for requesting and effectively using offsite assistance on site have been made, arrangements to accommodate State and local staff at the licensee's near-site Emergency Operations Facility have been made, and other organizations capable of augmenting the planned onsite response have been identified.

(2) Procedures have been established for licensee communications with State and local response organizations, including initial notification of the declaration of emergency and periodic provision of plant and response status reports.

(3) Provisions exist for prompt communications among principal response organizations to offsite emergency personnel who would be responding onsite.

(4) Adequate emergency facilities and equipment to support the emergency response onsite are provided and maintained.

(5) Adequate methods, systems, and equipment for assessing and monitoring actual or potential offsite consequences of a radiological emergency condition are in use onsite.

(6) Arrangements are made for medical services for contaminated and injured onsite individuals.

(7) Radiological emergency response training has been made available to those offsite who may be called to assist in an emergency onsite.

[45 FR 55409, Aug. 8, 1980, as amended at 47 FR 30235, July 13, 1982; 47 FR 40537, Sept. 15, 1982; 49 FR 27736, July 6, 1984; 50 FR 19324, May 8, 1985; 52 FR 42085, Nov. 3, 1987; 53 FR 36959, Sept. 23, 1988; 56 FR 64976, Dec. 13, 1991; 61 FR 30132, June 14, 1996]

§ 50.48 Fire protection.

(a) Each operating nuclear power plant must have a fire protection plan that satisfies Criterion 3 of appendix A of this part. This fire protection plan must describe the overall fire protection program for the facility, identify the various positions within the licensee's organization that are responsible for the program, state the authorities that are delegated to each of these positions to implement those responsibilities, and outline the plans for fire protection, fire detection and suppression capability, and limitation of fire damage. The plan must also describe specific features necessary to implement the program described above, such as administrative controls and personnel requirements for fire prevention and manual fire suppression activities, automatic and manually operated fire detection and suppression systems, and the means to limit fire damage to structures, systems, or components important to safety so that the capability to safely shut down the plant is ensured.[3] The licensee shall retain the fire protection plan and each change to the plan as a record until the Commission terminates the reactor license and shall retain each superseded revision of the procedures for three years from the date it was superseded.

(b) Appendix R to this part establishes fire protection features required to satisfy Criterion 3 of appendix A to this part with respect to certain generic issues for nuclear power plants licensed to operate prior to January 1, 1979. Except for the requirements of sections III.G, III.J, and III.O, the provisions of appendix R to this part shall not be applicable to nuclear power plants licensed to operate prior to January 1, 1979, to the extent that fire protection features proposed or implemented by the licensee have been accepted by the NRC staff as satisfying the provisions of appendix A to Branch Technical Position BTP APCSB 9.5–1[4] reflected in staff fire protection safety evaluation reports issued prior to the effective date of this rule, or to the extent that fire protection features were accepted by the staff in comprehensive fire protection safety evaluation reports issued before appendix A to Branch Technical Position BTP APCSB 9.5–1 was published in August 1976. With respect to all other fire protection features covered by appendix R, all nuclear power plants licensed to operate prior to January 1, 1979 shall satisfy the applicable requirements of appendix R to this part, including specifically the requirements of sections III.G, III.J, and III.O.

(c) All fire protection modifications required to satisfy the provisions of appendix R to this part or directly affected by such requirements shall be completed on the following schedule:

(1) Those fire protection features that involve revisions of administrative controls, manpower changes, and training, shall be implemented within 30 days after the effective date of this section and appendix R to this part.

(2) Those fire protection features that involve installation of modifications that do not require prior NRC approval or plant shutdown shall be implemented within 9 months after the effective date of this section and appendix R to this part.

(3) Those fire protection features, except for those requiring prior NRC approval by paragraph (c)(5) of this section, that involve installation of modifications that do require plant shutdown, the need for which is justified in the plans and schedules required by the provisions of paragraph (c)(5) of this section, shall be implemented before startup after the earliest of the following events commencing 180 days or more after the effective date of this section and appendix R to this part:

(i) The first refueling outage;

(ii) Another planned outage that lasts for at least 60 days; or

[3] Basic fire protection guidance for nuclear power plants is contained in two NRC documents:

• Branch Technical Position Auxiliary Power Conversion System Branch BTP APCSB 9.5–1, ''Guidelines for Fire Protection for Nuclear Power Plants,'' for new plants docketed after July 1, 1976, dated May 1976.

• Appendix A to BTP APCSB 9.5–1, ''Guidelines for Fire Protection for Nuclear Power Plants Docketed Prior to July 1, 1976,'' for plants that were operating or under various stages of design or construction before July 1, 1976, dated August 23, 1976.

Also see Note 4.

[4] Clarification and guidance with respect to permissible alternatives to satisfy Appendix A to BTP APCSB 9.5–1 has been provided in four other NRC documents.

''Supplementary Guidance on Information Needed for Fire Protection Evaluation,'' dated October 21, 1976.

''Sample Technical Specification,'' dated May 12, 1977.

''Nuclear Plant Fire Protection Functional Responsibilities, Administrative Control and Quality Assurance,'' dated June 14, 1977.

''Manpower Requirements for Operating Reactors,'' dated May 11, 1978.

A Fire Protection Safety Evaluation Report that has been issued for each operating plant states how these guidelines were applied to each facility and identifies open fire protection issues that will be resolved when the facility satisfies the appropriate requirements of Appendix R to this part.

(iii) An unplanned outage that lasts for at least 120 days.

(4) Those fire protection features that require prior NRC approval by paragraph (c)(5) of this section, shall be implemented within the following schedule: Dedicated shutdown systems—30 months after NRC approval; modifications requiring plant shutdown—before startup after the earliest of the events given in paragraph (c)(3) commencing 180 days after NRC approval; modifications not requiring plant shutdown—6 months after NRC approval.

(5) Licensees shall make any modifications necessary to comply with these requirements in accordance with the above schedule without prior review and approval by NRC except for modifications required by section III.G.3 of appendix R to this part. Licensees shall submit plans and schedules for meeting the provisions of paragraphs (c)(2), (c)(3), and (c)(4) within 30 days after the effective date of this section and appendix R to this part. Licensees shall submit design descriptions of modifications needed to satisfy section III.G.3 of appendix R to this part within 30 days after the the effective date of this section and appendix R to this part.

(6) In the event that a request for exemption from a requirement to comply with one or more of the provisions of Appendix R filed within 30 days of the effective date of this rule is based on an assertion by the licensee that such required modifications would not enhance fire protection safety in the facility or that such modifications may be detrimental to overall facility safety, the schedule requirements of paragraph (c) shall be tolled until final Commission action on the exemption request upon a determination by the Director of Nuclear Reactor Regulation that the licensee has provided a sound technical basis for such assertion that warrants further staff review of the request.

(d) Fire protection features accepted by the NRC staff in Fire Protection Safety Evaluation Reports referred to in paragraph (b) of this section and supplements to such reports, other than features covered by paragraph (c), shall be completed as soon as practicable but no later than the completion date currently specified in license conditions or technical specifications for such facility, or the date determined by paragraphs (d)(1) through (d)(4) of this section, whichever is sooner, unless the Director of Nuclear Reactor Regulation determines, upon a showing by the licensee, that there is good cause for extending such date and that the public health and safety is not adversely affected by such extension. Extensions of such date shall not exceed the dates determined by paragraphs (c)(1) through (c)(4) of this section.

(1) Those fire protection features that involve revisions of administrative controls, manpower changes, and training shall be implemented within 4 months after the date of the NRC staff Fire Protection Evaluation Report accepting or requiring such features.

(2) Those fire protection features involving installation of modifications not requiring prior approval or plant shutdown shall be implemented within 12 months after the date of the NRC staff Fire Protection Safety Evaluation Report accepting or requiring such features.

(3) Those fire protection features, including alternative shutdown capability, involving installation of modifications requiring plant shutdown shall be implemented before the startup after the earliest of the following events commencing 9 months or more after the date of the NRC staff Fire Protection Safety Evaluation Report accepting or requiring such features:

(i) The first refueling outage;

(ii) Another planned outage that lasts for at least 60 days; or

(iii) An unplanned outage that lasts for at least 120 days.

(4) Those fire protection features involving dedicated shutdown capability requiring new buildings and systems shall be implemented within 30 months of NRC approval. Other modifications requiring NRC approval prior to installation shall be implemented within 6 months after NRC approval.

(e) Nuclear power plants licensed to operate after January 1, 1979, shall complete all fire protection modifications needed to satisfy Criterion 3 of appendix A to this part in accordance with the provisions of their licenses.

(f) Licensees that have submitted the certifications required under § 50.82(a)(1) shall maintain a fire protection program to address the potential for fires which could cause the release or spread of radioactive materials (i.e., which could result in a radiological hazard).

(1) The objectives of the fire protection program are to—

(i) Reasonably prevent such fires from occurring;

(ii) Rapidly detect, control, and extinguish those fires which do occur and which could result in a radiological hazard; and

(iii) Ensure that the risk of fire-induced radiological hazards to the public, environment and plant personnel is minimized.

(2) The fire protection program must be assessed by the licensee on a regular basis and revised as appropriate throughout the various stages of facility decommissioning.

(3) The licensee may make changes to the fire protection program without NRC approval if these changes do not reduce the effectiveness of fire protection for facilities, systems, and equipment which could result in a radiological hazard, taking into account the decommissioning plant conditions and activities.

[45 FR 76610, Nov. 19, 1980, as amended at 53 FR 19250, May 27, 1988; 61 FR 39300, July 29, 1996]

§ 50.49 Environmental qualification of electric equipment important to safety for nuclear power plants.

(a) Each holder of or an applicant for a license for a nuclear power plant, other than a nuclear power plant for which the certifications required under § 50.82(a)(1) have been submitted, shall establish a program for qualifying the electric equipment defined in paragraph (b) of this section.

(b) Electric equipment important to safety covered by this section is:

(1) Safety-related electric equipment.[3]

(i) This equipment is that relied upon to remain functional during and following design basis events to ensure—

(A) The integrity of the reactor coolant pressure boundary;

(B) The capability to shut down the reactor and maintain it in a safe shutdown condition; or

(C) The capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guidelines in § 50.34(a)(1) or § 100.11 of this chapter, as applicable.

(ii) Design basis events are defined as conditions of normal operation, including anticipated operational occurrences, design basis accidents, external events, and natural phenomena for which the plant must be designed to ensure functions (b)(1)(i) (A) through (C) of this section.

(2) Nonsafety-related electric equipment whose failure under postulated environmental conditions could prevent satisfactory accomplishment of safety functions specified in subparagraphs (i) through (iii) of paragraph (b)(1) of this section by the safety-related equipment.

(3) Certain post-accident monitoring equipment.[4]

(c) Requirements for (1) dynamic and seismic qualification of electric equipment important to safety, (2) protection of electric equipment important to safety against other natural phenomena and external events, and (3) environmental qualification of electric equipment important to safety located in a mild environment are not included within the scope of this section. A mild environment is an environment that would at no time be significantly more severe than the environment that

[3] Safety-related electric equipment is referred to as ''Class 1E'' equipment in IEEE 323–1974. Copies of this standard may be obtained from the Institute of Electrical and Electronics Engineers, Inc., 345 East 47th Street, New York, NY 10017.

[4] Specific guidance concerning the types of variables to be monitored is provided in Revision 2 of Regulatory Guide 1.97, ''Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident.'' Copies of the Regulatory Guide may be purchased through the U.S. Government Printing Office by calling 202–275–2060 or by writing to the U.S. Government Printing Office, P.O. Box 37082, Washington, DC 20013–7082.